## NRR-PMDAPEm Resource

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| **From:** | Rankin, Jennivine |
| **Sent:** | Tuesday, January 17, 2017 3:50 PM |
| **To:** | Schrage, John L:(GenCo-Nuc) |
| **Subject:** | Clinton Power Station, Unit 1 - Request for Additional Information regarding the License Amendment Request to Revise TS 5.5.13 for Permanent Extension of Type A and Type C Leak Rate Test Frequencies (MF7290) |
| **Attachments:** | MF7290 RAI final.docx |

Mr. Schrage,

By letter dated January 25, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16025A182), as supplemented by letter dated March 31, 2016 (ADAMS Accession No. ML16076A120), Exelon Generation Company LLC (EGC, the licensee) submitted a license amendment request for Clinton Power Station, Unit 1. The proposed amendment would revise the technical specifications to reflect a permanent extension of Type “A” Integrated Leak Rate Testing and Type “C” Leak Rate testing frequencies.

The NRC staff has reviewed the information provided in the license amendment request and determined that additional information is required in order to complete its review.

A draft request for additional (RAI) was transmitted on July 7, 2016, and a clarification call was held on January 17, 2017. As agreed upon, please submit your response to the RAIs by March 3, 2017. If you wish to alter the date of your response, please contact me at (301) 415-1530.

Please treat this e-mail as formal transmittal of the RAIs.

Thanks,
Jennie

Jennie Rankin, Project Manager
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

**Hearing Identifier:** NRR_PMDA
**Email Number:** 3274

**Mail Envelope Properties** (Jennivine.Rankin@nrc.gov20170117155000)

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**From:** Rankin, Jennivine

**Created By:** Jennivine.Rankin@nrc.gov

**Recipients:**
"Schrage, John L:(GenCo-Nuc)" <John.Schrage@exeloncorp.com>
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REQUEST FOR ADDITIONAL INFORMATION

EXTENSION OF LEAK RATE TESTING

CLINTON POWER STATION, UNIT NO. 1

DOCKET NO. 50-461

1. Section 1 of Attachment 1 to the January 25, 2016, submittal (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16025A182) requests revision to Clinton Power Station, Unit 1 (CPS) Technical Specifications (TS) 5.5.13 to:

   > [a]dopt a more conservative allowable test interval extension of nine months, for Type A, Type B and Type C leakage rate tests in accordance with NEI [Nuclear Energy Institute] 94-01, Revision 3-A.

   In a letter dated June 25, 2008 (ADAMS Accession No. ML081140105), the Nuclear Regulatory Commission (NRC) staff approved NEI 94-01, "Industry Guideline For Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 2 (ADAMS Accession No. ML100620847). The NRC staff also identified in Section 3.1.1.2 that to extend Type A (i.e. integrated leak rate testing (ILRT)) beyond 15 years, the licensee must provide information demonstrating that there is an unforeseen emergent condition. Additionally, this 9-month extension is not to be used for routine scheduling or planning purposes. Given the requirement for demonstration of a non-routine emergent condition and that CPS operates on a 12-month refueling outage cycle, address the reasons and circumstances wherein the licensee expects to need such an extension.

2. The following apply to the internal events Facts & Observations (F&Os) reported in Attachment 1 to the March 31, 2016 supplement (ADAMS Accession No. ML16091A077):

   a. F&Os 1-3, 1-26 and 6-8 found that room heat-up calculations were not performed to support the exclusion of room ventilation from the PRA model. The resolutions of these F&Os discuss room heat-up calculations performed for the switchgear rooms and the residual heat removal (RHR) 'B' room. Address whether these are the only rooms in the plant that the cited F&Os apply to and whether there are no other risk-significant rooms where loss of ventilation could cause equipment failure.

   b. The resolutions of F&Os 1-26 and 6-8 address the rooms cited in the F&Os (e.g. the main control room and several emergency core cooling system (ECCS) rooms) using evaluations for other rooms and qualitative arguments. Address how the calculations performed for a particular room are applicable to other rooms. The justification can include information such as volume and heat load comparisons.

c. For F&O 1-4, related to supporting requirement (SR) IE-A6, there was no evidence that a systematic evaluation of initiating events due to multiple equipment failures and routine system alignments has been performed. Address whether this systematic evaluation was performed and discuss whether any new initiating events have been identified.

d. In F&O 1-14, related to SR IFSN-A6, the peer review team found no evidence that a systematic assessment of the effects of jet impingement, pipe whip, humidity, temperature, etc. on systems, structures and component s (SSCs) was performed. Describe the systematic assessment of these internal flooding effects and summarize its conclusions.

e. In F&O 1-17, related to SR IE-C3, it is stated that the basis for recovery actions for certain initiating event fault trees was not documented. Similarly, F&O 1-21, related to SR AS-B3, states that no evaluation for the ECCS pump operation at post-containment venting conditions was provided. The licensee's dispositions of these F&Os indicate that the basis is documented in various plant-specific reports. Provide a discussion of the basis for crediting the recovery actions mentioned in F&O 1-17 and for crediting ECCS operation after containment venting mentioned in F&O 1-21.

f. For F&O 1-24, which is related to SR SY-A18, identifies the lack of a concerted effort to identify accident conditions that could cause system failures. F&O 1-22, related to SR AS-B3, also raises a similar concern. Briefly describe the process followed to identify, review, and model accident conditions that could cause system failures.

g. For F&O 1-27, which is related to SR SY-A24 and DA-C15, it was found that plant-specific data was not analyzed to support crediting RHR repair in the probabilistic risk assessment (PRA) model. The resolution to this F&O states that the repair data are based on industry experience as allowed by SR DA-C15 and are judged to be reasonable. The NRC staff's endorsement of the 2009 American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard RA-Sa-2009 in Regulatory Guide (RG) 1.200, Revision 2, includes the exception for SR DA-C15, which states that industry experience should be used only if plant-specific experience is insufficient to estimate the failure to repair. Explain why the plant specific experience is insufficient to estimate the failure to repair.

h. For F&O 1-43, which is related to SR QU-D1, it is stated that a review of the "significant" cutsets and accident sequences as defined in the standard was not performed. The resolution states that the top cutsets and accident sequences now represent a larger percentage of the total core damage frequency (CDF) compared to the time of the peer-review, but does not address the concern in the F&O. Similarly, F&O 1-46, related to SR QU-D5, states that a review of the non-significant cutsets and accident sequences was not performed. The resolution for F&O 1-46 contains a reference to a licensee guidance document, but does not address the concern of the F&O. Address whether a review of significant cutsets and accident sequences, as defined in the 2009 ASME/ANS PRA Standard RA-Sa-2009, as well as non-significant cutsets and accident sequences was performed.

3. In the March 31, 2006 supplement, resolution of F&O 5-7 states:

   > [t]he [Electric Power Research Institute] EPRI [Human Reliability Analysis] HRA Calculator is now used to quantify the probabilities of the [Human Error Probabilities] HEPs in the Clinton model…The transition to using the EPRI HRA Calculator did not represent a methodology change...

   In addition, resolutions of all F&Os related to the human reliability (HR) supporting requirements (F&Os 1-31, 1-32, 1-33, 1-34, 2-16, 3-13, and 5-10) seem to indicate that many HRA related changes to the PRA model were performed in the 2011 PRA update and/or as a result of using the EPRI HRA Calculator.

   The 2009 ASME/ANS PRA Standard RA-Sa-2009 defines a PRA upgrade as,

   > …incorporation into a PRA model of a new methodology or significant changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences.

   Non-mandatory Appendix 1-A of the PRA Standard cites “a different HRA approach to human error analysis…” as a potential PRA upgrade. Based on Section 1-5, “PRA Configuration Control,” of the standard and RG 1.200, Revision 2, Regulatory Position 1.4, “PRA Development, Maintenance, and Upgrade,” a PRA upgrade must be peer reviewed.

   a. Justify why this model change is not considered a PRA upgrade requiring a focused-scope peer review. If this change qualifies as an upgrade, provide the results from of the focused-scope peer review addressing the associated F&Os and their resolutions.

   b. Provide an overview of all other changes in the internal events PRA model that occurred after the 2009 peer review, and clarify whether any of these changes qualify as a PRA upgrade that would require a focused-scope peer review.

4. One of the topical report use conditions identified in the safety evaluation (SE) approving the methodology in EPRI TR-1009325, Revision 2 (ADAMS Accession No. ML081140105) is that the average leak rate for the pre-existing containment large leak class (i.e., Class 3b) be assigned a value of 100 times the maximum allowable containment leakage rate ($L_a$) instead of the previously used value of 35 $L_a$. Due to the methodology used, the pre-existing drywell large leak also uses the new postulated leakage rate (i.e., 100 times the maximum allowable drywell leakage rate [DWLb]). These leakage rates (i.e., 100 La and 100 DWLb) represent a substantial increase from the values used for the previous one-time frequency extension request (ADAMS Accession No. ML030370524).

In Section 4.1.3 of Attachment 4 to the submittal, the licensee refers to the drywell bypass leakage rate test (DWBT) risk assessment methodology used for its previous one-time ILRT/DWBT extension request as the basis for the approach used. In the same section, the licensee discusses the impact of drywell leakage on containment over-pressurization and CDF based on the margin found in the results of the deterministic calculations documented in the previous one-time ILRT/DWBT frequency extension request. However, discussion of the impact of the increased DBWT leakage of 100 DWLb on the amount of Cesium Iodide (CsI) released is not provided. Therefore, address why the increased DWBT leakage of 100 DWLb does not change the assignment to the EPRI accident Class 3a and Class 3b shown in Table 4.6-1 of Attachment 4 to the submittal.

5. Similar to the previous one-time ILRT/DWBT frequency extension request, summarize the results of the sensitivity analysis including the probability of drywell failures assigned to small (Class 3a) or large (Class 3b) drywell bypass (DWB) leakage, using the current "as-found" DWBT leakage data for all Mark III containments. Provide the changes (i.e., delta) in large early release frequency (LERF), population dose, conditional containment failure probability, and impact on baseline LERF. The NRC staff notes that the sensitivity documented in Section 6.3 of Attachment 4 to the submittal which increases the probability values of the small (Class 3a) and large (Class 3b) DWB leakage by a factor of 10, does not appear to capture the corresponding probabilities determined based on the historical DWB leakage data using the Chi-square upper bound value.

6. Section 5.7 of Attachment 4 to the submittal, states that other external hazards, including high winds and tornadoes, external floods, transportation accidents, and nearby facility accidents were not considered because of their negligible contribution to overall plant risk. This conclusion was reached based on the CPS Individual Plant Examination for External Events (IPEEE) analysis. Since the IPEEE studies have not been recently updated, discuss the applicability of the IPEEE conclusions with regards to each of the above-mentioned hazards to the current plant configuration and operating experience, and taking into account updated risk studies and insights.

7. The licensee has discussed its Fire PRA in Section A.3.1 in Appendix A of Attachment 4 to the submittal. Address the following regarding the fire PRA:

    a. Address, preferably quantitatively such as through sensitivity analyses, whether the estimated fire CDF (6.0E-6/yr) and LERF (9.21E-7/yr) are bounding with respect to the current state-of-the-art for fire PRA considering all approved guidance since NUREG/CR-6850 was first issued.

    b. Address whether any "unapproved/unreviewed analysis methods" were employed in the current application of the Fire PRA.

8. Section 5.3 in Attachment 4 to the submittal, states that Class 7 sequences are impacted by the ILRT/DWBT interval extension. The statements appears to be inconsistent with arguments in Section 4.1.3, which explains that Class 7 sequences are not impacted by the requested extension. As Tables 5.2-2, 5.3-1 and 5.3-2 do not show any impact on

Class 7, clarify the treatment of Class 7 sequences.

9. Table 3.4.4-1 “CPS Type A ILRT History” of Attachment 1 to the submittal, provides the details of the historical ILRT “As-found Leakage Rate” and “As-Left Leakage Rate” values. Section 9.2.3 “Extended Test Intervals” of Nuclear Energy Institute (NEI) 94-01 Revision 2-A states that:

   > [i]n the event where previous Type A tests were performed at reduced pressure (as described in 10 CFR 50, Appendix J, Option A), at least one of the two consecutive periodic Type A tests shall be performed at peak accident pressure ($P_a$).

   Provide the actual ILRT test pressures employed during the two most recent Type A tests.

10. In Table 3.8.1-1 “NEI 94-01 Revision 2-A Limitations and Conditions” of Attachment 1 to the submittal, the licensee indicates that:

    > CPS will utilize the definition in NEI 94-01 Revision 3-A, Section 5.0. This definition has remained unchanged from Revision 2-A to Revision 3-A of NEI 94-01.

    Table 3.4.4-1 of Attachment 1 to the submittal provide the details of the historical ILRT Containment “As-found Leakage Rate” and “As-Left Leakage Rate” values. With respect to Table 3.4.4-1:

    a. Provide the definition of “performance leakage rate” used during the February 2008.

    b. From the Table 3.4.4-1 of ILRT results, it can be seen that the “Leakage Rate” increased by 22.9% [0.2708 “As found”/ 0.2204 “As Left”] between the time the ILRT of November 1993 was performed and when the ILRT of February 2008 was performed. Discuss the cause of the increase in containment leakage rates.

11. In Table 3.8.1-1 of Attachment 1 to the submittal, the licensee indicates that “…there no major modifications planned.” Section 9.2.4 of NEI TR 94-01, Rev. 2, indicates that Type A testing is required after major modifications to the containment or upon approval by the NRC the licensee may perform a short duration structural test of the containment. For minor modifications or modifications to the pressure boundary an LLRT was indicated. As the Unit 1 containment has been in service for approximately 28 years, provide a summary of all significant modifications to the Unit 1 containment and the subsequent post modification testing. The summary should discuss the extent to which actions were completed consistent with the NRC staff approval of NEI TR 94-01, Rev. 2 dated June 25, 2008 (ADAMS Accession No. ML081140105).

12. Table 3.6.5-1, “CPS Type B and C LLRT Program Implementation Review” of Attachment 1 to the submittal, identified components that were on extended intervals and have not demonstrated acceptable performance for Type C testing during the previous two outages (C1R14-2013, and C1R15-2015). Based on the footnotes for the table, address whether the ”As-found SCCM” for the three components in the table are

"As-found maximum SCCM" for those penetrations and the "As-left SCCM" for the same components are "As-left maximum SCCM" for the corresponding penetrations. If any of the "As-found SCCM" in Table 3.6.5-1 are "As-found minimum SCCM", address whether they are included in Table 3.6.4-1 "As-Found min path (SCCM)."

13. Like other Mark III containments, Unit 1 has an internal drywell interfacing with the surrounding primary containment through a suppression pool. When a loss-of-coolant accident occurs, the drywell atmosphere pushes down on the suppression pool displacing it past the weir wall into the drywell to the volume under the hydraulic control unit (HCU) floor. The surge of water and drywell atmosphere gas rises to the HCU floor and given the area restriction of the floor, a short term differential pressure is developed from below (wetwell space) to above the floor. When $P_a$ is calculated describe which region (i.e. above or below HCU floor) is credited. If the $P_a$ is not equal to or conservative relative to the peak pressure below the HCU floor, provide the peak calculated DBA-LOCA internal pressure in the wetwell space, justify the calculated $P_a$ value and the acceptability of performing ILRT/LLRT at a lower pressure than would occur in the wetwell space.